UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

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June 27, 1979

The Honorable John D. Dingell, Chairman
Subcommittee on Energy and Power
Committee on Interstate and Foreign Commerce
United States House of Representatives
Washington, D. C. 20515

Dear Mr. Chairman:

I appreciate the comments made in your letter to me of March 23, 1979 regarding the decision to require shutdown of five nuclear plants. The capability to shutdown nuclear power plants safely in the event of an earthquake is an important requirement in our regulations, and the decision reflected the NRC's concerns that this capability was not assured.

Your letter noted that witnesses testifying before other Congressional Committees on behalf of Stone and Webster argued that other safe alternative responses were available to the Commission. Three such responses were cited; reducing reactor power, establishing an earthquake watch and increasing personnel at the plants. The first alternative, reducing the authorized power output of the plants, would reduce the stored energy in the core but would not alter the temperatures and pressures of the systems. Hence, the stresses in the piping and equipment would not be lower and this alternative would not reduce the chance of an earthquake-caused accident. The second alternative, establishing an earthquake watch, is not a practical action. Current technology does not permit accurate forecasting of earthquakes, their location or their intensity. Even if a watch were successful, little, if anything, could be accomplished to improve the capability of the plant to safely withstand the event. The last alternative was to increase the number of plant personnel to provide additional capability to shutdown the reactor. It is not likely that additional personnel would materially alter the actions taken in the event of an earthquake. The operators at the control room can manually shutdown the plant at the onset of any earthquake, and the plants would be automatically shutdown in the event of earthquake-caused damage. Additional personnel might be helpful in recovering from earthquake-caused damage which jeopardized the ability to keep the reactor core safely cooled. However, the effectiveness of this alternative can not be readily established. In our view, none of these alternatives would have corrected the possible deficiency in the design of the facilities or improved the capability of the plants to withstand an earthquake.

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You also expressed concern about the possibility that similar errors may exist in other computer codes. On April 14, 1979 an Inspection and Enforcement Bulletin was dispatched to all power reactor facilities with an operating license or construction permit. This bulletin, a copy of which is enclosed, was issued for the express purpose of identifying whether similar errors might have been used in other applications.

Preliminary information in response to this bulletin indicates that a number of operating plants used some form of algebraic summation technique in the analysis of one or more systems (about two dozen plants total, half of which used the method extensively in design). As is noted in the bulletin, action plans are required to be developed to reevaluate and, as necessary, upgrade equipment where this method was employed.

This kind of problem appears to be limited to plants of an earlier vintage than plants presently under review. For some years now, the NRC staff has been implementing a program of code verification. Detailed requirements for this are specified in the Standard Review Plan and in the pertinent Regulatory Guides. These requirements are intended to prevent the kind of problem encountered in the Stone and Webster design analyses. I am enclosing a brief summary of the staff's code verification program.

You also expressed concern about the promptness of Stone and Webster in discovering the existence of this problem and reporting it to the Commission. As the enclosed chronology indicates, this problem surfaced as a result of review of a stress analysis on an overweight valve. For some period of time the existence of the code error, or at least its significance, was not evident. The chronology reflects the persistent efforts of the NRC staff to obtain information to permit the safety issues to be accurately defined so that necessary actions could be taken. We were, however, troubled by the length of time it took for this error to be discovered and corrective action taken. Therefore, a special inspection of this matter as to a possible 10 CFR Part 21 violation was conducted by the Office of Inspection and Enforcement in May 1979. We will provide you with a copy of that report when it becomes available.

Your final comment concerned the ability of the Commission to discuss a matter under consideration. The Commission is keenly aware of the possible ramifications of the Pillsbury case regarding this matter and appreciates your concern. Recently NRC's Office of General Counsel analyzed Pillsbury v. FTC, 354 F.2d 942 (5th Cir. 1966) and more recent cases following the Pillsbury doctrine (e.g., Koniag, Inc., Village of Uyak v. Andrus 580 F.2d 601 (D.C. Cir. 1978); D.C. Federation of Civil Associations v. Volpe, 459 F.2d 1231 (D.C. Cir. 1971) cert. denied 450 U.S. 1030 (1972)).

The Office of General Counsel informed the Commission that these cases stand for the proposition that agency officials must be quite circumspect in responding to Congressional questions addressing the merits of ongoing agency adjudications. If agency officials are forced to prejudge the merits of pending adjudication, or explain how and why the individual reached a decision in a case still before the official, the agency is taking a risk that its final order issued in the adjudication may be remanded to the agency for further proceedings because the litigants have not been afforded due process. A court may also order a remand if it determines that Congressional pressure has forced an agency to base a decision, in part, on factors which the agency would not ordinarily consider.

The Commission is also aware of the line of cases which holds that an administrative decision maker may be disqualified from further participation in an adjudicatory matter if he makes public statements which would give the appearance of having prejudged the issues in question (e.g., Association of National Advertisers, Inc. v. Federal Trade Commission, 460 F.Supp. 996 (D.D.C. 1978). Such cases, as well as the Pillsbury line, mandate that the Commission make every effort not to make statements which would create even the appearance of prejudgment.

Accordingly, while responding to its duty under Section 202 of the Atomic Energy Act of 1954 to keep the Congress fully and currently informed of Commission activities, the Commission has at the same time made every effort to avoid any problems that might stem from the Pillsbury doctrine.

I appreciate your concerns that the Commission not be in a position where it might appear that those future decisions of ours would be prejudiced. However, I have been advised that the Commission's appearances before Congress should not inhibit our ability to act on any future staff recommendations to us on this matter.

Sincerely,

Joseph M. Hendrie

Enclosures:
1. Summary of Code Verification Program
2. I&E Bulletin
3. Chronology

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Present State of Verification of Stress
Analysis Methods

Existing detailed requirements contained in pertinent Standard Review Plans and Regulatory Guides issued since the five plants were designed and approved have greatly reduced the chances that design errors of this type will take place. The Standard Review Plan sections and the Regulatory Guides which pertain to seismic analysis require a dynamic analysis, and provide for input time histories, ground response spectra, damping, modelling of structures, development of floor response spectra, and methods of combination of both spatial components and modal contributions. The Standard Review Plan also requires that applicants verify their dynamic analysis programs by comparison of results with those of other programs and with generally accepted solutions to benchmark problems.

To improve our confidence in computer results, the staff has for some time been in the process of establishing a standardized program for independently evaluating and verifying the quality of computer programs used for dynamic and static structural analysis of nuclear piping systems and components. This program consists mainly in the definition and solution of a set of standardized benchmark problems involving the analysis of a set of structures of progressively increasing complexity, representing typical piping system analyses

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as found in currently proposed or operating plants. Increased assurance of proper code verification will be provided by requesting applicants to provide solutions generated with their computer programs to these standardized benchmark problems, and comparing these responses with the benchmark solutions. Agreement or deviation of results will provide an index of the adequacy and quality of an applicant's analysis methods. This program will also provide the NRC with the capability to perform independent calculations to verify applicants' dynamic analyses for particular designs.

The following paragraphs elaborate on the past and present staff efforts in the area of stress analysis code review and verification.

In 1973, the staff realized that there was a proliferation of computer programs for stress analysis, all of which would be required to be examined in the process of licensing reviews. Due to the substantial number of plants under review at that time, it was decided that a generic program to review these computer programs should be instituted that would have two goals:

1. To provide independent in-depth verification of the capabilities of the programs claimed by the applicants in the SARs; and
2. To provide the staff with a list of acceptable computer programs that would reduce the review effort in at least one area.

In February 1974, an outline for a validation program was developed proposing that computer programs be evaluated and verified by means of benchmark problems and solutions. These benchmark problems were to be developed independently by the staff, and submitted to applicants requesting that they provide solutions to these problems. The acceptability of an applicant's computer program would be determined by the similarity of the applicant's solutions and the benchmark solutions.

In October 1974, a work scope entitled, "Piping Benchmark Problems" was issued for assistance from a national laboratory in generating the benchmark solutions. This work scope described the requirements for such a program, and a preliminary list of problems suitable to be used as benchmarks. The Brookhaven National Laboratory in Upton, New York, was chosen to provide the required solution. In Fiscal Year 1975, the actual benchmark problems were selected by the NRC staff and BNL personnel, and computer programs that were to be used for generating the solutions were chosen and verified. Actual generation of benchmark solutions was begun in FY-1976. The computer program chosen for this effort was the program SAP-IV (Structural Analysis Program), developed at the University of California at Berkeley in the early 1970's and widely available.

Two reports detailing five benchmark problems and solutions were published in December 1977 (BNL-NUREG-21241-RS and BNL-NUREG-23645), and a draft request for information became available in January of 1978. The benchmark problems in these reports pertain to linear elastic structures and range from a simple structure under static loading to a two-loop primary piping system compiling a reactor vessel, steam generators, pumps and supports, subjected to earthquake motion. Additional benchmark problems have since been developed which pertain to elastic structures involving gaps (a non-linear problem). Other problems are being developed which include newer techniques, such as multiple support excitation, and preliminary efforts have been made in developing benchmarks for inelastic piping analysis.

In the course of licensing reviews, the NRC staff has required description and verification of structural programs since the early 1970's, and formalized these requirements in the Standard Review Plan published in 1975, (Section 3.9.1). Applicants submitted verification solutions which were based on simple benchmark problems only. The Piping Benchmark Program was designed to complement and expand these requirements and provide additional verification. However, methods of analysis of nuclear power plants for structural response under seismic and other loading conditions, which were the basis for these computer programs and were used in the design of early power plants (1968), have been presented in the open literature since the late 1960's.

Applicants have also provided descriptions and verifications of their computer programs in the form of topical reports. One such topical report was submitted in 1976 by the Westinghouse Electric Co. titled: "Documentation of Selected Westinghouse Structural Analysis Computer Codes" (WCAP-8252). These programs and solutions were reviewed as thoroughly as possible without actually performing computer calculations, except for one program which involved a nonlinear analysis. The benchmark problem which the applicant submitted was reviewed under the Piping Benchmark Program by the BNL, by generating an independent solution to the same problem and confirming the applicant's results. (This problem will be incorporated in our standard list of benchmark problems.) Duke Power Co. also submitted verification of its method for structural analysis. The results by this applicant were also verified independently by BNL by running the same problems under the Piping Benchmark Program. A final report on this method will be published in the near future. Other analyses have been verified independently by the staff, and we are presently performing an evaluation and verification of the design techniques of certain component support members.

Related to the Benchmark Program is a much more general computer program evaluation project sponsored by the Armed Forces, and conducted by a group called the Interagency Software Evaluation Group (ISEG). The NRC staff is represented on this group. The objective of the group is to evaluate in depth the capabilities of some of the very large structural computer programs, such as ADINA, used nationwide.

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CHRONOLOGY TABLE

PIPE STRESS ANALYSIS ISSUE

Note: This chronology represents information available to the NRC as of March 17, 1979. It does not necessarily fully or accurately reflect the actual sequence of events which occurred prior to the March 8, 1979 meeting.

10/2/78	Stone and Webster notified the Beaver Valley Unit 1 Station Manager of an error discovered in the original hand calculated stress analysis of some safety injection lines. The error was discovered while evaluating the impact of correcting the weight on 14 safety injection system check valves. Since this was, technically, a deviation from the Final Safety Analysis Report, it was to be referred to the Station Safety Committee. To that end, the Station Superintendent asked for more specific information on the error*.
10/13/78	At a meeting held at the Beaver Valley site between Duquesne Light Company and Stone and Webster representatives, additional information on the error was provided but more specifics were requested by DLC*.
10/23/78	Stone and Webster provided DLC more information. The Station Superintendent asked for additional clarification and was told Stone and Webster personnel would be at the site the next week*.
10/26/78	During a site visit, Stone and Webster informed DLC that one safety injection line would actually be significantly overstressed. DLC then made a prompt telephone notification to Region I of the Office of Inspection and Enforcement*.

*These entries provided from memory by Duquesne Light Company representative on March 17, 1979.

10/26/78	Prompt report LER 78-053/01P to NRC Region I via telecon from Duquesne Light Company. Reported information received from Stone and Webster that hand calculation errors resulted in stress levels above ANSIB 31.1, 1967 but only in one case of six flow paths.
10/27/78	Daily Report by Region I to I&E headquarters included as a reportable occurrence - inadequate piping supports during review of safety injection pipe s ress analysis by the A/E (S&W), several points ı the 6-inch and smaller piping were found to be inadequately supported. In the event of safety injection system operation during a DBE, 5 points could exceed the code allowable stress. A design change for safety injection piping supports will be accomplished prior to unit startup in mid-November.
10/27/78	Written interim LER submitted by Duquesne Light Company. DLC characterized the errors reported by Stone and Webster as resulting from a hand calculation method of analysis.
10/31-11/3/78	IE Inspection 50-334/78-30 - Region I followup on 24 hour report. Inspector raised a number of questions including: What assurance can be given to show that the calculational error applies only to the six points in question? To only the Safety Injection system? To only the Beaver Valley facility?
11/9/78	Second interim LER submitted by Duquesne Light Company indicates that the original report was erroneous. The line stresses were thought to have been hand calculated only, when in fact they were subsequently computer calculated and found acceptable. DLC also indicated that a full report on the situation was in preparation by Stone and Webster.

11/14-17/78	IE Inspection 50-334/78-33 - Region I inspectors followup but no information available onsite.
11/16/78	Region I Daily Report indicated a rereview by A/E found that the previously reported condition was erroneous and that no inadequately supported piping existed, a full report of the situation is being prepared by the A/E and a followup to the LER will be submitted by the Licensee to NRC.
11/30/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/01/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/04/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/05/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/06/78	LER 78-53/01T-0 was submitted to NRC by licensee. Conclusion was that "corrective action has been reviewed, approved and satisfactorily completed". The report based on information supplied by Stone and Webster attributes the pipe overstress to differences between stresses analyzed by PSTRESS code and those done by the chart method. It mentions differences between PSTRESS and NUPIPE codes in force summation but does not elaborate on them. It concludes that PSTRESS used methods acceptable for Beaver Valley Unit 1 generation plants. It states that Reg. Guide 1.92 issued in December 1974 established for facilities docketed after April 1975 more conservative techniques for intramodal combinations of generalized loadings. The report states that analysis showed that only one safety injection system pipe required modification - the addition of one snubber and the redesign of one support. The attachment to this LER provided additional historical information as follows:

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Duquesne Light Company reported in an attachment to the December 6, 1978 LER 78-53/01T-0 that to generate data needed for installation of a net positive suction head modification to the Beaver Valley Unit 1 safety injection system, they (Stone and Webster) decided to "code in" the six inch SI lines into a currently used computer program (NUPIPE). DLC indicated original design used the PSTRESS code. No results of an analysis at this stage were reported by DLC to NRC.

Subsequent to the above activity the attachment states the Beaver Valley Power Station was notified by a vendor that check valves in SI system were actually heavier than used in design at construction stage. This increased weight was used as input to the above NUPIPE model and found not to "affect" the piping design. The Architect Engineer (Stone and Webster) also concluded that the hanger designs need not be changed as a result of using the correct (heavier) weight for these valves. However errors were said to have been discovered in the hand calculation method. It was determined that piping analysis showed local overstress at several anchors but no overstress in "the pipe" alone.

Per attachment to LER 78-53/01T-0, a more thorough evaluation was initiated to determine if "any other annulus piping" originally designed by the chart (hand calculation) method was overstressed.

Per attachment to LER 78-53/01T-0, licensee found that SI lines had been "as-built" reviewed in 1974 and that two of the six lines had been (at that time) coded into PSTRESS (not just hand calculation method). The PSTRESS code was re-run using the correct valve weights and resulted in acceptable pipe stresses.

Also per attachment to LER-78-53/01T-0, licensee states "The models run in PSTRESS and NUPIPE are geometrically similar; however, the mass distribution and support stiffness are different. Further, the method of force summation (intra-modal) is different. NUPIPE utilizes more conservative techniques for intra-modal combinations of generalized loadings.

These newer techniques arose following establishment of Beaver Valley Unit No. 1 design criteria. In December, 1974, the USNRC published Regulatory Guide 1.92, applicable to facilities docketed after April, 1975, which required the use of the more conservative combinations. The PSTRESS methods used were accepted dynamic analysis techniques for Beaver Valley Unit 1 generation plants, and is the basis for all computerized Category I pipe stress analysis performed".

(It is NRC understanding that results were unsatisfactory on two of three lines, but snubber and support modifications on one line reduced the overstress on the second line such that no modifications on that line were necessary.)

The pre December 6, 1978 review of annulus seismic piping was limited to lines that had been previously analyzed using the hand calculation method (2-1/2 inch to 6 inch lines). 103 lines were identified, 55 were reviewed and found acceptable. Licensee noted that PSTRESS results were still available for 48 of the 103 lines from the 1974 as built review and were "acceptable".

Licensee notes its Engineering Department is "continuing a review of the architect-engineer findings".

12/11/78 Followup calls to site by the IE inspector to seek additional information.

Region I IE inspector telephoned NRR Licensing Project Manager to obtain a contact for informal discussion of technical questions.

12/12/78 Region I Daily Report - Further review of in-containment SI system piping supports identified one line requiring support modification, attributed to an error in original design calculations.

12/14/78 Regional inspector was telephoned by NRR individual who was designated as contact. Preliminary technical discussion was held about potential problems.

12/18-20/78	IE Inspection 50-334/78-34 - Region I followup on 12/6 LER. During this inspection, the inspector reviewed the detailed report submitted to the licensee by A/E and discussed the results of that review with representatives of the licensee and A/E.
12/22/78	Region I inspector discussed with NRR individuals via telephone questions he had as a result of discussions he had with S&W on 12/18-20/78. The NRC individuals involved determined that there was a possible problem.
1/18/79	Region I mailed to IE Headquarters a memorandum requesting that information be forwarded to NRR for review. The memo defined concerns to include: 1. Reconciliation of the differing analysis results to assure that the design methods used are neither incorrect nor unconservative. 2. The need for further licensee review of piping potentially affected by any incorrect or nonconservative calculation.
1/23/79	The IE Inspector provided copy of the 01/18/79 memorandum to Licensing Project Manager.
About 2/2/79	Discussion between IE inspector and NRR project manager determined that a formal transfer of lead responsibility between I&E and NRR had not been made of the 01/18/79 memorandum to NRR.
2/2/79	A formal request for DOR's Engineering Branch support (TAC form) was prepared by the project manager.
2/5/79	IE inspector was informed by IE:HQ that telephone discussion had established that NRR was working on the problem and that a formal transfer of lead to NRR would be made.
3/1/79	During a conference call to DLC and S&W, a computer run was requested for DOR review. Since S&W corporate policy was not to provide such proprietary data, a meeting was set up for S&W to bring in a computer run for DOR review at Bethesda.

3/8/79 A technical meeting was held between DLC, S&W, and the NRC staff to discuss and review the PIPESTRESS and NUPIPE codes. The NRC approached the review with the belief that the two codes were acceptable and that some modeling or input problem created the results in question. It was revealed that the PIPESTRESS code used an algebraic summation of seismic loads which in the absence of a detailed time history analysis, gave unconservative results in the seismic stresses. Management was immediately informed and a management level meeting arranged with DLC and S&W.

3/8/79 A management level meeting was held with DLC and S&W to arrange for immediate review of the Beaver Valley pipe stress analyses. Commitments were requested of S&W to identify the systems and plants involved, the inadequacies expected and the reanalysis to confirm safe operation. No definitive information was available at that time. DLC was requested to have its plant safety committee review the situation.

3/9/79 Numerous staff meetings were held at Bethesda to scope the problem with respect to the effects if a seismic event were to occur. Telecons were made to S&W on the schedule of commitments for further information on Beaver Valley. The other utilities identified by S&W as having plants with the same problem were notified. These plants and utilities were: Fitzpatrick, Power Authority of the State of New York; Maine Yankee, Maine Yankee Atomic Power Company; Surry 1 and 2, Virginia Electric and Power Company.

The Chairman was advised. Three staff members were sent to Boston to provide immediate review and analysis of results. DLC sent eight people to Boston to assist in expediting the review.

3/9/79 In view of the problems and with the Offsite Safety Review Committee concurrence, the Beaver Valley Unit 1 was placed in hot standby for the weekend by DLC to await further analyses from S&W.

3/10/79 Staff meetings continued as pieces of information were fed back from Boston. The I&E Duty Officers were advised of actions. The NSSS vendors for the plants were contacted to assure no other codes for pipe stress during that period used the same algebraic approach. A DOR Assistant Director was sent to Boston to provide management review and coordination. S&W's computer was dedicated full time to these stress calculations and extended work hours for data reduction was instituted for S&W staff. NRC options were explored and draft materials developed to support appropriate action based on the technical results becoming available on Beaver Valley.

3/11/79 Early S&W reanalysis results on Beaver Valley runs indicated problems with pipes as well (originally thought only supports). Licensees' top management was contacted to assure action underway by all plants to identify inadequacies and obtain reanalyses of stresses in all affected safety systems.

3/12/79 Additional information from DOR staff in Boston confirmed pipe stresses above allowable and unacceptable.

Arrangements were made to brief the Commission on this matter. All the licensees were notified of a pending decision.

3/13/79 In view of the safety significance of this matter as discussed above, the Director of the Office of Nuclear Reactor Regulation proposed to the Commission that the public health and safety requires that the present suspension of operation of the facility should be continued: (1) until such time as the piping systems for all safety systems have been reanalyzed for earthquake events to demonstrate conformance with General Design Criterion No. 2 using a piping analysis computer code which does not contain the error discussed above, and (2) if such reanalysis indicates that there are components which deviate from applicable ASME Code requirements, until such deviations are rectified. The Commission concurred in the NRR Director's decision.

Prior to the NRC final decision to order the plants shutdown, the Beaver Valley Offsite Safety Review Committee recommended the facility be placed in cold shutdown based on the data and analysis recieved from S&W. The DLC ordered the plant shutdown.

3/14/79 The licensees confirmed by telecon that the Orders were received and provided times when each facility would be in cold shutdown. All facilities will be at or below 200°F by 10:40 p.m. on March 15, 1979 in conformance with the Order.

Subsequently all affected licensees were notified by telephone that the Orders were executed and that a copy would be transmitted by facsimile.

3/16-17/79 Meetings were held with Stone and Webster with the Utilities to discuss acceptable methods of analysis for interim and long term fixes of the piping and supports.